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**U.S. NUCLEAR REGULATORY COMMISSION**

# STANDARD REVIEW PLAN

**Office of Nuclear Reactor Regulation**

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## DRAFT STANDARD REVIEW PLAN SECTION 15.0.2

## REVIEW OF ANALYTICAL COMPUTER CODES

**REVIEW RESPONSIBILITIES**

Primary - Lead Branch for Accident or Transient

Secondary - Other Branches with Responsibility for Accident or Transient

### I. AREAS OF REVIEW

In order to establish a licensing basis, licensees must analyze transients and accidents per the requirements of 10 CFR 50.34, 10 CFR 50.46, and where applicable, per NUREG-0737, "Clarification of TMI Action Plan Requirements." These accidents and transients are described in Chapter 15 of the Standard Review Plan (SRP) (NUREG-0800) [Ref. 1]. This section of the SRP describes the review process and acceptance criteria for analytical models and computer codes used by licensees to analyze accident and transient behavior. The purpose of the review is to verify that the evaluation model is adequate to simulate the accident under consideration. This includes methods to estimate the uncertainty in the calculation, as in the case of a best estimate loss of coolant accident (LOCA); or to ensure that the results of the analysis are demonstrably conservative, as in the case of a LOCA in Appendix K to 10 CFR Part 50.

The guidance in this section should be applicable to most of the transients and accidents described in the SRP. Appendices that provide specific information for specific classes of

accidents and transients will be developed in the future. Guidance to the industry is proposed in Draft Regulatory Guide DG-1096, “Transient and Accident Analysis Methods” [Ref. 2].
The areas to be reviewed are:

**1. Documentation**

The development of a computer code for use in reactor safety licensing calculations requires a substantial amount of documentation. This documentation includes/covers: (a) the evaluation model, (b) the accident scenario identification process, (c) the code assessment, (d) the uncertainty analysis, (e) a theory manual, (f) a user manual, and (g) the quality assurance program.

**2. Evaluation Model**

An evaluation model is the calculation framework for evaluating the behavior of the reactor coolant system during a postulated accident or transient. It includes one or more computer programs and other information necessary for application of the calculation framework to a specific transient or accident, such as mathematical models used, assumptions included in the programs, a procedure for treating the program input and output information, specification of those portions of the analysis not included in the computer programs, values of parameters, and other information necessary to specify the calculation procedure. Evaluation models are sometimes referred to as a licensing methodology.

**3. Accident Scenario Identification Process**

The accident scenario identification process is a structured process used to identify and rank the reactor component and physical phenomena modeling requirements based on (a) their importance to acceptable modeling of the scenario and (b) their impact on the figures of merit for the calculation (e.g., peak cladding temperature and maximum and average cladding oxidation thicknesses). It is also used to identify the key figures of merit or acceptance criteria for the accident.

**4. Code Assessment**

The code assessment provides a complete assessment of all code models against applicable experimental data and/or exact solutions in order to demonstrate that the code is adequate for analyzing the chosen scenario.

**5. Uncertainty Analysis**

Uncertainty analyses are performed to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest when the code is used in a licensing calculation. Examples of safety parameters are Peak Cladding Temperature (PCT), cladding oxidation thickness, Departure From Nucleate Boiling Ratio (DNBR), and Critical Power Ratio (CPR).

**6. Quality Assurance Plan**

The quality assurance plan covers the procedures for design control, document control, software configuration control and testing, and error identification and corrective actions used in the development and maintenance of the evaluation model. The program also ensures adequate training of personnel involved with code development and maintenance, as well as those who perform the analyses.

## II. ACCEPTANCE CRITERIA

The acceptance criteria are based on meeting the requirements of the regulations in 10 CFR Part 50 that govern the evaluation models for the specific accident under consideration (e.g., 10 CFR 50.46 for a LOCA). The specific criteria are as follows:

### 1. Documentation

The submittal must identify the specific accident scenarios and plant configurations for which the codes will be used. The evaluation model documentation must be scrutable, complete, unambiguous, accurate, and self contained. Consistent nomenclature must be used throughout the entire model documentation. Any referenced material must be readily available from a technical library. Copies of any referenced documents that are not readily obtainable from a technical library, including proprietary reports, must be included with the documentation or provided upon request. The code documentation must be sufficiently detailed that a qualified engineer can understand the documentation without recourse to the originator as required of any design calculation that meets the design control requirements of 10 CFR Part 50, Appendix B, and the documentation requirement in 10 CFR Part 50, Appendix K. The documentation must include the responses to requests for additional information, sorted according to the review issue so that it is easy to follow the entire review history for a single issue. The documentation must include:

A. An overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation.

B. A complete description of the accident scenario including plant initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and systems and/or component interactions that influence the outcome of the accident.

C. A complete description of the code assessment comprising a description of each assessment test, why it was chosen, success criteria, diagrams of the test facility that show the location of instrumentation that is used in the assessment, a code model nodalization diagram, and all code options used in the calculation.

D. A determination of the code uncertainty for a sample plant accident calculation.

E. A theory manual that is a self-contained document and that describes: (a) field equations, (b) closure relationships, (c) numerical solution techniques, (d) simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods, (e) pedigree or origin of closure relationships used in the code, and (f) limits of applicability for all models in the code.

F. A user manual that provides: (a) detailed instructions about how the computer code is used, (b) a description of how to choose model input parameters and appropriate code options, (c) guidance about code limitations and options that should be avoided for particular accidents, components, or reactor types, and (d) if multiple computer codes are used, documented procedures for ensuring complete and accurate transfer of information between different elements of the evaluation model.

G. A quality assurance plan that describes the procedures and controls under which the code was developed and assessed, and the corrective action procedures that are followed when an error is discovered.

## 2. Evaluation Model

Models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident under consideration. The chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view. The degree of imprecision that is allowed in the models will ultimately be determined by the amount of uncertainty that can be tolerated in the calculation. Models that cause non-physical predictions to the extent that misinterpretation of the calculated results or trends in the results may occur, are not acceptable. For Appendix K LOCA analyses, emergency core cooling system (ECCS) evaluation models must meet the specific requirements contained in 10 CFR 50, Appendix K.

## 3. Accident Scenario Identification Process

The purpose of the accident scenario identification process is to identify and rank the reactor component and physical phenomena modeling requirements based on: (a) their importance to the modeling of the scenario and (b) their impact on the figures of merit for the calculation. The accident scenario identification process must be a structured process. It must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario. The models that are present in the code and their degree of fidelity in predicting physical phenomena must be consistent with the results of this process. For example, if the accident scenario identification process determines that a certain physical phenomenon is important to the scenario under consideration, the code must have a relatively accurate model for that phenomenon and a detailed assessment of that model must be provided. Phenomena that have lower ranking may be represented by models with larger inherent uncertainty.

## 4. Code Assessment

Assessments of all code models must be provided. All assessments must be performed with the frozen version of the evaluation model that has been submitted for review. Assessments performed with other versions of the evaluation model are not acceptable because even “small” changes to the evaluation model can have unintended consequences on calculation results that were thought to not be impacted by the changes.

Separate effects testing must be performed to demonstrate the adequacy of the physical models to predict physical phenomena that were determined to be important by the accident scenario identification process. Separate effects testing must also be used to determine the uncertainty bounds of individual physical models.

Integral effects testing must be performed to demonstrate that the interactions between different physical phenomena and reactor coolant system components and subsystems are identified and predicted correctly.

Assessments against both separate effects tests and integral effects tests must be performed with the code. All models need to be assessed over the entire range of conditions encountered in the transient or accident scenario. Assessments must also compare code predictions to analytical solutions, where possible, to show the accuracy of the numerical methods used to solve the mathematical models. Code options used in the assessment calculations must be the same as those used in plant accident calculations.

A scaling analysis must be performed that identifies important non-dimensional parameters related to geometry and key phenomena. Scaling distortions and their impact on the code assessment must be identified and evaluated in the assessment. Calculations of actual plant transients or accidents can be considered, but only as confirmatory supporting assessments. This is because the data available from plant instrumentation is generally not detailed enough to support code assessment. The assessment cases must compare code predictions to important measured variables in order to show that good predictions of one test variable do not result from compensating errors. Assessments must include a description of all assessment cases, specific models that are being assessed in each case, and acceptance criteria used. Acceptance criteria must be supported by quantitative analysis whenever possible.

ECCS evaluation models must include a specific assessment to meet the 10 CFR Part 50,
Appendix K criteria. Small break ECCS evaluation models must also meet the assessment requirements of TMI Action Item II.K.3.30 where applicable.

## 5. Uncertainty Analysis

The uncertainty analysis must address all important sources of code uncertainty, including the mathematical models in the code and user modeling such as nodalization. The major sources of uncertainty must be addressed consistent with the results of the accident sequence identification process. When the code is used in a licensing calculation, the combined code and application uncertainty must be less than the design margin for the safety parameter of interest. The analysis must include a sample uncertainty evaluation for a typical plant application.

In some cases, bounding values are acceptable for input parameters described in SRP sections or Regulatory Guides and are used for plant operating conditions such as accident initial conditions, set points, and boundary conditions.

## 6. Quality Assurance Plan

The code must be developed and maintained under a quality assurance program that meets the requirements of 10 CFR Part 50, Appendix B.

# III. <u>REVIEW PROCEDURES</u>

## 1. Assignment of Review Responsibilities

The lead reviewer for the evaluation model should perform an assessment of the submittal in order to determine what areas of reviewer expertise are required to begin the review process.  Evaluation models often contain complex mathematical and physical models that encompass multiple scientific and engineering disciplines.  The lead reviewer should identify qualified personnel and their areas of review responsibility.  Reviews often require personnel with highly specialized areas of expertise which may not be available in the NRR technical staff.

## 2. Acceptance Review

An acceptance review is a preliminary process of assessment of the completeness and apparent acceptability of quality of the documents submitted in order to commit NRC to a detailed review of the application.  The reviewers of the code or codes comprising the evaluation model should perform an acceptance review consistent with the guidance provided in NRR Office Letter No. 803, Revision 3, “License Amendment Review Procedures” [Ref. 3].  The reviewers should ensure that all areas discussed in the section titled Documentation are addressed in the topical report submitted by a licensee or vendor.  The documentation is not required to have the same section titles or organization as described above, but all of the relevant subject matter must be included.  The material must be sufficiently detailed to permit the reviewers to begin a review without immediately requesting additional documentation in order to understand the licensing submittal.  The results of the acceptance review should be documented in a letter to the organization submitting the evaluation model for review.  Submittals that do not contain the required material should also be processed in accordance with Revision 3 of NRR Office Letter No. 803 (Ref. 3).  The reviewers should document what material is missing from the submittal.  This information should then be provided in the letter to the submitting organization so that the material can be revised to meet acceptance review requirements for any future submittals.  After a successful completion of the acceptance review, a detailed review of the code documentation may proceed.

## 3. Detailed Review

### a. Documentation

The reviewers should review the documentation to determine if (i) all documentation listed in Section II.1 above has been provided, (ii) the evaluation model overview provides an accurate roadmap of the evaluation model documentation, (iii) all documentation is accurate, complete, and consistent and, (iv) all symbols and nomenclature have been defined and consistently used.

The reviewers should confirm that the theory manual is a self-contained document and that it describes the field equations, closure relationships, numerical solution techniques, and simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods.  The reviewers should also confirm that the theory manual identifies the pedigree or origin of closure relationships used in the code and the limits of applicability for all models in the code.

The reviewers should confirm that the user manual provides guidance for selecting or calculating all input parameters and code options.  The guidance must be clear and unambiguous.  As a result of the code development or review process many code options may be determined to be inappropriate for specific licensing calculations.  The reviewers should confirm that the guidance in the manual specifies the required and acceptable code options for the specific licensing calculations.  The reviewers should also confirm that required input settings are hardwired into the input processor so that the code stops with an error message if the required input is not provided or if the input is not within an acceptable range of values.  The reviewers should confirm that computer codes that are used for multiple accidents and transients include guidelines that are specific to each transient or accident.  Code assessment cases can be examined to ensure that the modeling used in the assessment cases is consistent with the user guidelines.

**b. Evaluation Model**

The reviewers should determine whether the mathematical modeling and computer codes used to analyze the transient or accident have been previously reviewed and accepted. For changes to previously approved models, the reviewers can limit their review to the new material if they determine that there is nothing new that will invalidate the previous approval, including the range of applicability for the analysis method.  Otherwise the entire model must be reviewed. For a new model that has not been previously reviewed, the reviewers initiate an evaluation of the entire analytical model.

The reviewers should determine if the physical modeling described in the theory manual and
contained in the mathematical models is adequate to calculate the physical phenomena influencing the accident scenario for which the code is used.  A scenario will have a set of governing physical phenomena that drive the results of the calculation.  The key physical phenomena, including constitutive equations needed for model closure, must be defined for the calculation being performed by the code.  Physical phenomena that are important for one accident scenario may not be important for a different accident scenario.  The key physical phenomena can also be specific to a particular plant design.

The mathematical equations that comprise an evaluation model can be characterized as being either a field equation or as a constitutive or closure relationship.  Field equations are a set of rigorously derived equations that contain no approximations other than the initial assumptions used in deriving the equations.  The range of applicability of the field equations is limited only by the validity of the assumptions used in their derivation.  An example of a set of field equations is the set of fluid transport equations for mass, momentum, and energy that are derived from macroscopic balances of these quantities. Although these equations are mathematically exact, the equations contain more unknown quantities than there are equations.  Some of these unknown quantities are the equation of state, the stress tensor, and the heat flux.  In order to be able to solve the field equations the unknown quantities must be expressed in terms of the known quantities from the field equations.  The equations that relate the unknown quantities to the known quantities are constitutive or closure relationships.  These equations are often models or approximations that are much more restrictive in their range of validity than the set of field equations that they are used with.  For example, using Newton’s law of cooling as a closure relationship for heat flux from a heat structure to a fluid will limit the application to model situations where radiation heat transfer is not significant even though the field equations are valid.  The reviewer must therefore ensure that the field equations of the evaluation model are adequate to describe the set of physical phenomena that occur in the accident and ensure that the closure relationships are valid over the full range of conditions encountered during the accident.

The modeling must be consistent with the results of the accident scenario identification process in that there must be models for all important phenomena in the accident scenario.  Components and physical phenomena that are identified as being important in the accident scenario identification process must be modeled with a high degree of fidelity.  Phenomena of lower importance may be represented by less accurate models.

The reviewers should determine if the simplifying assumptions and assumptions used in the averaging procedure are valid for the accident scenario under consideration. Simplifying assumptions and averaging are often applied to detailed physical and mathematical modeling to obtain simplified mathematical models that can be solved more readily and with less computational effort.  Examples of common simplifications are incompressible flow models, one-dimensional flow models, common two-phase flow models such as the homogeneous equilibrium model (HEM), drift flux, and the two-fluid model, and simplified reactor kinetics models such as point kinetics or one-dimensional kinetics.  Even models commonly thought of as detailed models usually contain simplifying assumptions and averaging procedures applied to first-principles models. Reviewers should confirm that justifications are provided for all simplifications, assumptions, and averaging.

The reviewers should confirm that the level of detail in the model is equivalent to or greater than the level of detail required to specify the answer to the problem of interest. For example, a one-dimensional flow model can not provide information about the velocity profile in the vicinity of a pipe bend or the degree of thermal stratification in a horizontal pipe.  A detailed, three-dimensional flow model would be needed to provide this type of information.

The reviewers should confirm that the equations and derivations are correct.  There must be sufficient text to adequately describe the derivation, including all assumptions and equa-tions.  The derivations must be sufficiently detailed to allow the reviewers to understand the logical progression of steps involved in the derivation.  Simplifying assumptions must have a technical justification and a range of validity associated with them.

Models that are typically used in nuclear reactor analysis are highly phenomenological and/or empirical in nature.  They are either proposed using physical or engineering judgement based on observations of experimental data or derived using averaging procedures applied to detailed first-principle models.  These models often represent processes that occur on length and time scales that are too small to be resolved in the computation or processes that we do not have sufficient understanding to model from first principles.  These models require closure relationships based on information from experimental measurements or detailed first-principle calculations. The reviewers should ensure that the range of validity of the closure relationships is specified and is adequate to cover the range of conditions encountered in the accident scenario. This is especially true of empirical correlations which are derived directly from experimental data without recourse to any physical modeling.

Code developers sometimes take a well known correlation and modify it in an attempt to expand the applicability of the original correlation. They name the new correlation the “modified well known” correlation even though it may have little to do with the original correlation and the new correlation may not even be valid for the conditions that the original correlation was developed for. The reviewer must ensure that the code documentation provides assessment and verification of the new correlation over its full range of application. The reviewer should not rely on any assessment of the original correlation as being applicable to the new correlation.

In most applications, especially those with a large number of processes and parameters, it is difficult, if not impossible, to design test facilities that preserve total similitude between the experiment and the nuclear power plant. Therefore, optimum similarity criteria are identified and scaling rationales developed for selecting existing data or designing and operating experimental facilities. The reviewers should confirm that the similarity criteria and scaling rationales are based on the important phenomena and processes identified by the accident scenario identification process and appropriate scaling analyses.

The reviewers should confirm that scaling analyses were conducted to ensure that the data and the models will be applicable to the full scale analysis of the plant transient. Scaling compromises that are identified must be addressed in the bias and uncertainty evaluation. The experimental data base must be demonstrated to be sufficiently diverse, so that the expected plant specific response is bounded and that the evaluation model calculations are comparable to the corresponding tests in non-dimensional space. This demonstration allows extending the conclusions relating to code capabilities, drawn from assessments comparing calculated and measured test data to the prediction of plant specific transient behavior.

### c. Accident Scenario Identification Process

The accident scenario identification process is required in order to determine the needed modeling and assessment requirements for the code. The accident scenario identification process is also needed to identify and rank the reactor component and physical phenomena modeling requirements based on their importance to acceptable modeling of the scenario and their impact on the figures of merit for the calculation. This process is highly dependent on the type of reactor and the accident scenario of interest.

Often a single computer code is used to analyze multiple accident or transient classes. A separate accident scenario identification description is needed for each accident or transient class for which the code is to be used in order to describe the accident progression and dominant physical phenomena for that particular accident. This description must explicitly reference code models and assessment cases that are specifically applicable to each scenario to avoid confusion when the same code is used for multiple accident scenarios.

The processes and phenomena that evaluation models should simulate are found by examining experimental data and experience, as well as code simulations related to the specific scenario.  Independent techniques to accomplish the ranking include expert opinion, selected calculations, and decision making methods.

The reviewers should confirm that the process used for accident scenario identification is a structured process.  The reviewers should confirm that the description of each accident scenario provides a complete and accurate description of the plant initial and boundary conditions and the accident progression.  The specified initial and boundary conditions must comply with values required by regulations or specified as acceptable in a Regulatory Guide or SRP Section that covers the accident.  The reviewers should confirm that the dominant physical phenomena influencing the outcome of the accident are correctly identified and ranked.  The reviewers should keep in mind that the initial phases of the accident scenario identification process can rely heavily on expert opinion and can be subjective.  Therefore, iteration of the process, based on experimentation and analysis, is important.

One example of an acceptable structured process is the phenomena identification and ranking table (PIRT) process, which is described in NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident" [Ref. 4].  The process is also described in a series of papers in the journal *Nuclear Engineering Design* [Ref. 5].

**d. Code Assessment**

The reviewers should confirm that the code assessment adequately covers all of the important code models and the full range of conditions encountered in the accident scenarios.  The assessment must be consistent with the accident scenario identification process in that all models must have assessment commensurate with their importance and required fidelity.  The reviewers should also verify that all assessment cases were performed with a single version of the evaluation model.

The reviewers should confirm that the numerical solution conserves all important quantities.  Even when the mathematical equations conserve mass, momentum, and energy, the numerical method used to solve the equations may not conserve any of these quantities.  The reviewers should also confirm that all code options that are to be used in the accident simulation are appropriate and are not used merely for code tuning.

The reviewers should ensure that all code closure relationships based in part on experimental data or more detailed calculations have been assessed over the full range of conditions encountered in the accident scenario by means of comparison to separate effects test data.  Scaling analyses may be needed to demonstrate that the assessment results apply to the full-scale plant accident conditions.  Even closure relationships such as equations of state and material properties, which are based on interpolating functions, need to be assessed against standards for the properties.  For example, a relatively small error in thermodynamic properties such as phasic densities as a function of temperature and pressure may cause a larger error when propagated to a neutron kinetics model in a main steamline break or an anticipated transient without scram (ATWS) calculation.

The reviewers should ensure that integral test assessments properly demonstrate physical and code model interactions that are determined to be important for the full size plant accident scenarios. The integral test will usually not be full-scale, and therefore will contain scaling distortions. These distortions can affect both local and overall elements of the analysis, such as two-phase flow regime transitions and global dynamic response of the test facility, when compared to the full scale plant. The reviewers should ensure that the documentation contains comparisons of all important experimental measurements with the code predictions in order to expose possible cases of compensating errors. Such errors may result in good predictions of key parameters, derived from poor predictions of contributing parameters. These cases are often indicative of tuning the code to an integral experiment. One example of this is described in NUREG/CR-5249 [Refs. 4 and 5], in which key parameters, namely large break LOCA (LBLOCA) reflood peak cladding temperatures were predicted well, even though an important contributing factor, the core void fraction during reflood, was not predicted accurately.

In the case of LOCA evaluation models, specific assessment test cases are required in order to meet the requirements of 10 CFR 50, Appendix K. Specific test cases are also specified in the TMI action items for PWR small break LOCA (SBLOCA) evaluation models. The reviewers should confirm that these assessments have been performed where applicable.

The reviewers should refer to published literature for sources of assessment data for specific phenomena, accident scenarios, and plant types. These include the ECCS Compendium [Ref. 6], NEA validation data documents [Refs. 7, 8, and 9], and International Standard problems.

### e. Uncertainty Analysis

The reviewers should confirm that the method for calculating uncertainty contains all important sources of uncertainty and that a sample uncertainty calculation for a prototypical plant gives a reasonable estimate of the calculation uncertainty. The reviewers should confirm that the accident scenario identification process was used in identifying the important sources of uncertainty.

The reviewers should confirm that sources of code uncertainty have been addressed. These include uncertainties in theoretical models or closure relationships determined from comparison to separate effects tests, uncertainties due to scaling of the basic models and closure relationships, and uncertainties due to plant nodalization and solution techniques. The reviewers can sometimes determine whether the code uncertainty is reasonable by applying a simple analytical model. This can be done if the dominant contribution to uncertainty is confined to a small number of models. In other cases, detailed audit calculations may be needed to confirm the estimate of uncertainty in a calculation.

The reviewers should confirm that sources of calculation uncertainties have been addressed, including uncertainties in plant model input parameters for plant operating conditions (e.g., accident initial conditions, set points, and boundary conditions). Calculation uncertainties are specific to a given licensing calculation. Therefore, it is sometimes acceptable to use bounding values in the licensing calculation for these types of uncertainties.

The reviewers should confirm that the uncertainties in the experimental data base have been addressed. These uncertainties arise from such items as measurement errors and experimental distortions. For separate effects tests and integral effects tests, the reviewers should confirm that the differences between calculated results and experimental data for important phenomena have been quantified for bias and deviation. The reviewers should confirm that data sets and correlations with experimental uncertainties that are too large when compared to the requirements for evaluation model assessment are not used.

When the code is used in a licensing calculation, the reviewers should confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest in the calculation.

For best estimate LOCA analyses, uncertainty determination description and guidance are described in NUREG/CR-5249 [Ref. 4], Regulatory Guide 1.157, and in Appendix A to DG-1096 [Ref. 2]. In these examples, the uncertainty analyses discussed have the ultimate objective of providing a singular statement of uncertainty with respect to the 10 CFR 50.46 acceptance criteria. This singular uncertainty statement is accomplished when the individual uncertainty contributions are determined (see Regulatory Guide 1.157, Ref. 10).

For other Chapter 15 events, a complete uncertainty analysis is not required. However, in most cases the SRP guidance is to use “suitably conservative” input parameters. This suitability determination may involve a limited assessment of biases and uncertainties. The individual uncertainty (in terms of range and distribution) of each key contributor is determined from the experimental data, input to the nuclear power plan model, and the effect on appropriate figures of merit evaluated by performing separate calculations. The figures of merit and devices chosen must be consistent.

The NRC has developed the Code Scaling, Applicability, and Uncertainty (CSAU) methodology for code uncertainty evaluation. The CSAU process has been demonstrated for LBLOCA [Refs. 4 and 5] and boiling water reactor ATWS [Ref. 11].

### f. Quality Assurance Plan

The reviewers should confirm that the evaluation model was developed and maintained under a quality assurance program that meets the requirements of 10 CFR Part 50, Appendix B. As a minimum, the program must address design control, document control, software configuration control and testing, and corrective actions. The reviewers should confirm that the quality assurance program documentation includes procedures that address all of these areas. The reviewers may conduct an audit of the implementation of the code developer’s quality assurance program.

The reviewers should confirm that independent peer reviews were performed at key steps in the evaluation model development process. The peer review team should include programmers, developers, end users, and independent members with recognized expertise in relevant engineering and science disciplines, code numerics, and computer programming. Expert peer review team members, who were not directly involved in the evaluation model development and assessment, can enhance the robustness of the evaluation models. Further, they can be of value in identifying deficiencies that are common to large system analysis codes.

### 4. Independent Analysis

The reviewers may perform independent analyses in order to determine or confirm the importance of key phenomena and evaluate the impact of uncertainties in these phenomena on the key figures of merit in the plant calculation. The reviewers may want an independent determination of the importance of a high-ranking phenomenon, or to determine if a phenomenon not ranked as high by the licensee must be so ranked. Independent analysis may also be used to confirm uncertainty estimates. The Office of Research should be consulted, as needed, to accomplish these audits.

## IV. EVALUATION FINDINGS

The reviewers should document all findings in an evaluation model safety evaluation report (SER) that either accepts the evaluation model for the intended use or rejects the evaluation model. Acceptance may be subject to limitations determined during the review. If the evaluation model is to be accepted, it must be clearly demonstrated that the model is getting the right answer for the right reasons. There must be no evidence of compensating errors or arbitrary code tuning that produces the desired result for a single parameter. Any errors found in the documentation or issues related to the technical adequacy of the model must be addressed through the review process. The review process and the results of communications with the code submitter must also be documented as appropriate in the SER to provide a traceable history of the review process. Restrictions and limitations on use of the evaluation model must be explicitly documented in the SER. The SER must document areas that were reviewed, the method of review, and the findings in each area. The SER must also document areas that were not reviewed and provide the reasons for their omission.

The reviewers should determine if the evaluation model was previously reviewed and if the evaluation model previously had restrictions on its use or on the use of certain models in the codes. These previous restrictions must be explicitly addressed in the SER. The current review of the evaluation model may result in the imposition of new restrictions to be placed on its use. These restrictions must be explicitly identified. The applicant's documentation must also be revised to reflect these restrictions on use of the code. SER restrictions that have not been explicitly incorporated into evaluation model documentation have been determined to not be legally enforceable. Only statements in the approved description of the evaluation model are legally enforceable.

## V. NOMENCLATURE AND DEFINITIONS

All definitions are in the context of the objectives of this SRP section and may not be generic to other uses.

| | |
|---|---|
| Acceptance Review | An initial review of a submittal performed to ensure that sufficient information is included for the review to conduct the review. |
| Accidents (Transients) meet the analyses. | In this SRP section, accidents and transients refer to events that are defined in Chapter 15 of NUREG-0800 (Ref. 1) to be analyzed to the requirements of the General Design Criteria (GDC), except for fuel assembly misloading event and all radiological consequence |

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|---|---|
| ATWS | Anticipated transient without scram. |
| Codes | Calculational procedures that compose an evaluation model. |
| Code Options | User controlled options that control which procedures, models, correlations, etc., the code uses when performing calculations. |
| Code Tuning | The adjustment of parameters or options included in the code to achieve a predetermined result of code run. |
| Compensating Errors | A set of errors that, combined, mask the effect of the individual errors. |
| Correlation Effort | The change in one parameter as a result of a change in another parameter |
| CFR | Code of Federal Regulations |
| Closure Relationships | Equations and correlations required to close the field equations so that they may be solved. They relate unknown quantities to the variables of the field equations. They may include physical correlations of transport phenomena such as equations that relate the shearing stresses to the rate of strain (the velocity field). |
| Constitutive Equations | Equations and correlations required to close the field equations so that they may be solved. They relate unknown quantities to the variables of the field equations. They may include physical correlations of transport phenomena such as equations that relate the shearing stresses to the rate of strain (the velocity field). |
| CPR | Critical Power Ratio - The ratio of assembly power at which critical heat flux occurs to the actual assembly power. |
| CSAU | Code Scaling, Applicability, and Uncertainty - A process to determine the applicability, scalability, and uncertainty of a computer code in simulating an accident or other transient. A PIRT process is normally imbedded within a CSAU process. See Reference 1. |
| DG-1096 | Draft Regulatory Guide DG-1096, “Transient and Accident Analysis Methods” [Ref. 2] |
| DNBR | Departure from nucleate boiling ratio |
| ECCS | Emergency core cooling system |

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|---|---|
| Empirical | Derived from experimental data without recourse to physical modeling. |
| Evaluation Model (EM) | Calculational framework for evaluating the behavior of the reactor system during a postulated Chapter 15 event, which includes one or more computer programs and all other information needed for use in the target application. |
| Field Equations | Equations that are solved to determine the transport of mass, energy and momentum throughout the system. |
| Figures of Merit | Quantitative standards of acceptance that are used to define acceptable answers for a safety analysis (e.g., DNBR limits and fuel temperature limits). |
| frozen | The condition whereby the analytical tool(s) and associated facility input decks remain unchanged (and under configuration control) throughout a safety analysis thereby ensuring traceability of, and consistency in, the final results. |
| GDC | General Design Criteria - Design criteria described in Appendix A to 10 CFR Part 50. |
| HEM | Homogeneous Equilibrium Model - An analytical model for two-phase flow that assumes (1) both phases move at the same velocity, (2) the fluid is in thermal equilibrium, and (3) the flow is isentropic and steady. |
| Integral Effects Test | An experiment in which the primary focus is on the interactions between parameters and processes. |
| LB | Large break |
| LOCA | Loss of coolant accident |
| Model | (without “evaluation” modifier) - Equation or set of equations that represents a particular physical phenomenon within a calculational device. |
| PCT | Peak Cladding Temperature - The maximum fuel element cladding temperature. |
| Phase | State of matter involved in transport process, usually liquid or gas. |
| PIRT | Phenomena Identification and Ranking Table - May refer to a table, or to a process depending on context of use. The process relates to determining the relative importance of phenomena (and/or physical processes) to the behavior of a nuclear power plant following the initiation of an accident or other transient. A PIRT table is a listing of the results of application of the process. |

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| QA | Quality assurance |
| RAI | Request for additional information - A documented request, usually in the form of questions, sent by the NRC to the submitter to obtain more information on areas under review. |
| Roadmap | A document used to facilitate navigation through a larger and complex set of documentation. |
| SB | Small break |
| Scaling | The process in which the results from a subscale facility (relative to a nuclear power plant) and/or the modeling features of a calculational device are evaluated to determine the degree to which they represent a nuclear power plant. |
| Scaling Distortions | Errors introduced into the data as a result of scaling the experimental facility. |
| Scenario | Time sequence of events |
| Sensitivity Studies | The term is generic to several types of analyses; however, the definition of most interest here relates to those studies associated with the PIRT process and used to determine the relative importance of phenomena/ processes. This may also involve analysis of experimental data that are a source of information used in the PIRT process. |
| Separate Effects Test | An experiment in which the primary focus is on a *single* parameter or process. |
| SER | Safety Evaluation Report - A report by the NRC that evaluates a submittal and either accepts or rejects the proposals in the submittal. |
| SRP | Standard Review Plan - Acceptable plan for NRC reviewers in NUREG-0800. |
| Transients (Accidents)<br><br>fuel analyses. | In this SRP, accidents and transients refer to those events that are defined in Chapter 15 of NUREG-0800 to be analyzed to meet the requirements of the General Design Criteria (GDC), except for the assembly misloading event and all radiological consequence |
| Uncertainty | There are two separate but related definitions of primary interest:<br><br>a. The inaccuracy in experimentally derived data typically generated by the inaccuracy of measurement systems.<br>b. The inaccuracy of calculating primary safety criteria or related figures of merit typically originating in the experimental data and/or assumptions used to develop the analytical tools. The |

analytical inaccuracies are related to approximations in solving the equations and constitutive relations.

User Manual — A document that includes modeling guidelines for the accident under consideration, procedures for selecting code inputs, and procedures for transferring information between different pieces of the evaluation model.

## VI. REFERENCES

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2. USNRC, "Transient and Accident Analysis Methods," Draft Regulatory Guide DG-1096, December 2000.[2]

3. NRR Office Letter No. 803, Revision 3, "License Amendment Review Procedures," December 30, 1999. (Electronic copies are available under ADAMS Accession Number ML993550418 in NRC's Public Electronic Reading Room, which can be accessed through the NRC's web site, <WWW.NRC.GOV> .)

4. B.E. Boyak et al., "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," NUREG/CR-5249, USNRC, December 1989.[1]

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6. "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, December 1988.[1]

7. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations, NEA/CSNI/R(93)14, September 1993.[3]

8. "Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients," Committee on the Safety of Nuclear Installations, NEA/CSNI/R(96)17, July 1996.

9. "CSNI Code Validation Matrix of Thermo-Hydraulic Codes for LWR LOCA and Transients," Committee on the Safety of Nuclear Installations, CSNI Report 132, March 1987.[3]

10. USNRC, "Best-Estimate Calculations of Emergency Core Cooling System Performance," Regulatory Guide 1.157, May 1989.[2]

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[1] Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; (telephone (703)487-4650). Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or (800)397-4209; fax (301)415-3548; email is PDR@NRC.GOV.

[2] Single copies of regulatory guides, both active and draft, and draft NUREG documents may be obtained free of charge by writing the Reproduction and Distribution Services Section, OCIO, USNRC, Washington, DC 20555-0001, or by fax to (301)415-2289, or by email to <DISTRIBUTION@NRC.GOV>. Copies of certain guides and many other NRC documents are available electronically on the internet at NRC's home page at <WWW.NRC.GOV> in the Reference Library. Documents are also available through the Public Electronic Reading Room (NRC's ADAMS document system, or PERR) at the same web site.

[3] Copies are available at <http://www.nea.fr/html/nsd/reports/csnirepindex.html> .

11. W. Wulff et al., " Uncertainty Analysis of Suppression Pool Heating During an ATWS in a BWR-5 Plant: An Application of the CSAU Methodology Using the BNL Engineering Plant Analyzer," NUREG/CR-6200, USNRC, March 1994.[1]